UNITED STATES
**NUCLEAR REGULATORY COMMISSION**
WASHINGTON, D.C. 20555-0001

September 28, 2009

Mr. Peter P. Sena III
Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 2 - RELIEF REQUEST NO. 2-TYP-3-RV-02 REGARDING THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE CASE N-729-1 EXAMINATION REQUIREMENTS (TAC NO. ME0349)

Dear Mr Sena:

By letter dated December 30, 2008, FirstEnergy Nuclear Operating Company (licensee) submitted a request for authorization of a proposed alternative to American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-729-1 volumetric and surface examination coverage requirements for certain Beaver Valley Power Station, Unit No. 2 (BVPS-2) reactor vessel head penetrations for the remainder of the BVPS-2 third 10-year inservice inspection (ISI) interval, scheduled to end in 2018. Specifically, the licensee defined an alternate examination zone below the J-groove weld.

The Nuclear Regulatory Commission (NRC) staff has concluded that compliance with Section 50.55a(g)(6)(ii)(D) of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, and that the proposed alternative provides reasonable assurance of structural integrity. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff authorizes the proposed alternative for BVPS-2 for the third 10-year ISI interval, or until the reactor vessel head is replaced, whichever occurs first.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Beaver Valley Project Manager, Nadiyah Morgan, at (301) 415-1016.

Sincerely,

Nancy L. Salgado

Nancy L. Salgado, Chief
Plant Licensing Branch 1-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-412

Enclosure:
As stated

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**UNITED STATES**
**NUCLEAR REGULATORY COMMISSION**
**WASHINGTON, D.C. 20555-0001**

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING THE 10-YEAR INSERVICE INSPECTION PLAN INTERVAL

FOR RELIEF REQUEST NO. 2-TYP-3-RV-02

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 2

DOCKET NO. 50-412

## 1.0 INTRODUCTION

By letter dated December 30, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090020385), FirstEnergy Nuclear Operating Company (licensee) submitted a request for authorization of a proposed alternative to American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-729-1 volumetric and surface examination coverage requirements for certain Beaver Valley Power Station, Unit No. 2 (BVPS-2) reactor vessel head penetrations for the remainder of the BVPS-2 third 10-year inservice inspection (ISI) interval, scheduled to end in 2018. Specifically, the licensee defined an alternate examination zone below the J-groove weld.

## 2.0 REGULATORY EVALUATION

The ISI of the ASME Code, Section XI, Class 1, 2 and 3 components shall be performed in accordance with the requirements of Section XI, "*Rules for Inservice Inspection of Nuclear Power Plant Components*," of the ASME Code and applicable editions and addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission.

Enclosure

Pursuant to 10 CFR 50.55a(g)(4), throughout the service life of a pressurized-water reactor (PWR), components which are classified as ASME Code Class 1, 2, and 3 must meet the requirements, except design and access provisions and pre-service examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry and materials of construction of the components. Further regulations, under 10 CFR 50.55a(g)(4)(i), require that the ISI of components and system pressure tests conducted during the first 10-year ISI interval and subsequent intervals shall comply with the requirements in the latest edition and addenda of the ASME Code, Section XI, incorporated by reference in paragraph (b) of 10 CFR 50.55a on the date, 12 months prior to the start of the 120-month ISI interval subject to the limitations and modifications listed herein. The Section XI, ASME Code of Record for the third 10-year ISI interval at BVPS-2, which is scheduled to conclude in 2018, is the 2001 Edition through the 2003 Addenda.

10 CFR 50.55a(g)(6)(ii) states that the Commission may require the licensee to follow an augmented ISI program for systems and components for which the Commission deems that added assurance of structural reliability is necessary. 10 CFR 50.55a(g)(6)(ii)(D) requires augmented ISI of reactor vessel head penetration nozzles of PWRs in accordance with ASME Code Case N-729-1, subject to the conditions specified in paragraphs (2) through (6) of 10 CFR 50.55a(g)(6)(ii)(D).

Pursuant to 10 CFR 50.55a(a)(3), proposed alternatives to the requirements of 10 CFR 50.55a(g) may be used when authorized by the Commission if: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Under 10 CFR 50.55a(a)(3), the licensee requests relief from the requirements of 10 CFR 50.55a(g)(6)(ii)(D) for the third 10-year ISI interval of BVPS-2.

## 3.0 TECHNICAL EVALUATION

### 3.1 System/Component Affected

Control rod drive mechanism (CRDM) nozzle penetration numbers 1, 3, 5-34, 37, 38, 41-43, 45, 46, 48, 49, and 54-65, designated as Item Number B4.20, "UNS [unified numbering system] N06600 nozzles and UNS N06082 or UNS W86182 partial-penetration welds in head," in Table 1 of Code Case N-729-1.

### 3.2 Applicable Code Requirements

The Code of Federal Regulations 10CFR50.55a(g)(6)(ii)(D)(1) requires that examinations of the reactor vessel head be performed in accordance with ASME Code Case N-729-1 (Reference 8.1), subject to the conditions specified in paragraphs 10CFR50.55a(g)(6)(ii)(D)(2) through (6).

Paragraph 2500 of Code Case N-729-1 states, in part:

"*... If obstructions or limitations prevent examination of the volume or surface required by Fig. 2 for one or more nozzles, the analysis procedure of Appendix I shall be used to demonstrate the adequacy of the examination volume or surface* *for each such nozzle. If Appendix I is used, the evaluation shall be submitted to the regulatory authority having jurisdiction at the plant site."*

3.3 Licensee's Basis for Request

The bottom end of all of the BVPS-2 reactor vessel head CRDM penetrations are externally (outside diameter or "OD") threaded, internally (inside diameter or "ID") tapered, and have an ultrasonic corner shadow zone produced by the thread relief, precluding ultrasonic or eddy current data acquisition in a zone extending up approximately 1.45 inches from the bottom of each nozzle. For the majority of the penetrations, these geometric limitations reduce the inspectable distance from the bottom of the J-groove weld fillet to the top of the thread relief to some value less than the required coverage dimension "a" shown in Figure 2 of Code Case N-729-1.

Therefore, due to this physical hardship, relief is required for some CRDM penetrations that can not meet the required inspection coverage dimension "a" shown in Figure 2 of Code Case N-729-1.

During the BVPS-2 fall 2003 refueling outage, [the licensee] obtained examination coverage data on all 65 CRDM penetrations in the reactor vessel head. This information was used to support [the licensee's] previous NRC Order Relaxation Request regarding examination coverage below the J-groove weld, which was approved by the NRC [by letter dated August 2, 2004 (ADAMS Accession Number ML041950374)]. However, the issuance of 10 CFR 50.55a(g)(ii)(D) . . . on September 10, 2008 required implementation of [ASME] Code Case N-729-1 with NRC conditions by December 31, 2008.

The licensee supported RR 2-TYP-3-RV-02 with a stress analysis and deterministic fracture mechanics analysis. The plant-specific stress analysis demonstrated that the hoop and axial stresses remain below 20 kilo-pounds per square inch (ksi) over the entire region outside the alternative examination zone defined by the licensee's proposed alternative. The stress analysis was provided for NRC staff review. The plant-specific fracture mechanics analysis demonstrated that a potential axial crack in the unexamined zone will not grow to the toe of the J-groove weld prior to the next scheduled examination. The licensee noted that because previous primary water stress-corrosion cracking (PWSCC) had been found in the BVPS-2 head penetrations, volumetric re-inspection of the head was required each refueling outage. The licensee's analysis noted, for all cases, the crack growth predictions show greater than 4 years of full power operations required to grow the postulated flaw to the toe of the weld.

By letter dated July 29, 2003 (ADAMS Accession No. ML032130380), the licensee noted that while performance of dye penetrant testing may be possible in lieu of eddy current or ultrasonic testing to cover the area of missed inspection coverage, compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The presence of thermal sleeves in the vast majority of the CRDM penetrations prohibits dye penetrant testing of the tapered ID surface of the tube. Dye penetrant testing of threaded surfaces, like the tube OD, is difficult due to physical restraints and the need to properly clean the surface to provide accurate test results. As a result, performing dye penetrant testing on the bottom nozzle area would require thermal sleeve removal, extensive manpower, and would result in significant radiation exposure to personnel. The radiation exposure is estimated to be in excess of one hundred person-rem, without a compensating increase in the level of quality or safety.

3.4 Licensee's Proposed Alternative

The licensee proposes to perform ultrasonic examination of each reactor pressure vessel (RPV) head CRDM penetration nozzle (i.e., nozzle base material) for a distance equal to "a" above the J-groove weld on the uphill slope, as defined by Figure 2 of Code Case N-729-1, and to the minimum required inspection distances below the J-groove weld on the downhill slope as identified in Table 1. For all other penetrations, the required examination coverage dimension "a" reflected in Figure 2 of Code Case N-729-1 will be met or exceeded.

Table 1: BVPS-2 CRDM Nozzle Minimum Required Inspection Coverage

| Penetration No. | Minimum Inspection Coverage Below the J-Groove Weld Toe on the Downhill Side (in) | Penetration No. | Minimum Inspection Coverage Below the J-Groove Weld Toe on the Downhill Side (in) |
|---|---|---|---|
| 1 | 1.44 | 29 | 0.88 |
| 2 | 1.68 | 30 | 1.00 |
| 3 | 1.32 | 31 | 1.40 |
| 4 | 1.52 | 32 | 1.24 |
| 5 | 1.24 | 33 | 1.12 |
| 6 | 1.26 | 34 | 0.92 |
| 7 | 1.20 | 37 | 0.88 |
| 8 | 1.12 | 38 | 0.84 |
| 9 | 1.40 | 41 | 0.92 |
| 10 | 1.40 | 42 | 0.92 |
| 11 | 1.08 | 43 | 0.96 |
| 12 | 1.40 | 45 | 0.88 |
| 13 | 1.36 | 46 | 0.80 |
| 14 | 1.32 | 48 | 0.88 |
| 15 | 1.20 | 49 | 0.88 |
| 16 | 1.44 | 54 | 0.96 |
| 17 | 1.26 | 55 | 0.80 |
| 18 | 1.12 | 56 | 0.72 |
| 19 | 1.19 | 57 | 0.92 |
| 20 | 1.00 | 58 | 0.40 |
| 21 | 1.32 | 59 | 0.76 |
| 22 | 1.08 | 60 | 0.68 |
| 23 | 1.20 | 61 | 0.68 |
| 24 | 1.28 | 62 | 0.48 |
| 25 | 1.20 | 63 | 0.68 |
| 26 | 1.04 | 64 | 0.88 |
| 27 | 1.24 | 65 | 0.60 |
| 28 | 1.12 | | |

3.5 NRC Staff's Evaluation

The NRC staff's review was based on 10 CFR 50.55a(a)(3)(ii) which states that:

> Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The specific regulatory requirements for which relief is requested are defined in 10 CFR 50.55a(g)(6)(ii)(D)(3), which states in part: "Instead of the specified 'examination method' requirements for volumetric and surface examinations in Note 6 of Table 1 of Code Case N-729-1, the licensee shall perform volumetric and/or surface examination of essentially 100 percent of the required volume or equivalent surfaces of the nozzle tube, as identified by Figure 2 of ASME Code Case N-729-1." The extent of the examination of the nozzle tube in question is determined by the incidence angle, Θ, and the distance "a" below the J-groove weld, as defined in Figure 2 of N-729-1, "a = 1.5 in. (38mm) for incidence angle, Θ, ≤ 30 deg and for all nozzles ≥ 4.5 in. (115 mm) OD or 1 in. (25 mm) for incidence angle, Θ, ≥ 30 deg; or to the end of the tube, whichever is less." The licensee has identified 53 CRDM penetration nozzles, in Table 1 above, for which this inspection coverage is not physically obtainable with ultrasonic inspection.

The licensee has shown a physical and radiological hardship which would be incurred in order to be within compliance with the specified requirements. The NRC staff finds that a physical hardship exists due to the inability of ultrasonic or eddy current inspection to effectively scan the bottom end of each CRDM penetration as each nozzle is threaded on the OD and internally tapered. While dye penetrant inspection would be an option for the licensee, the inspection would require manual application in a high radiation area. Further, additional setup work would require additional accumulation of dose for each nozzle. Therefore, the NRC staff finds that the radiological dose required to perform the additional inspection would be a significant radiological hardship for the limited additional inspection coverage.

The NRC staff then compared the regulatory requirement to the proposed alternative to ensure that given this hardship, compliance with the regulation did not provide a compensating increase in the level of quality and safety. The NRC staff reviewed the licensee's basis for the proposed alternative through a stress and fracture mechanics analysis.

The NRC staff's review of the stress analysis was based on the degradation phenomenon of concern being PWSCC. PWSCC typically initiates in the areas of the highest tensile stress in susceptible materials, such as UNS N06600 material, and propagates in a controlled fashion in response to time, environment (i.e. temperature) and stress intensity. The NRC staff reviewed the licensee's stress analysis and conclusions by comparison of the licensee's supporting data for various nozzle angles, the conservative analysis performed to support Materials Reliability Program Report, MRP-95R1, "Generic Evaluation of Examination Coverage Requirements for Reactor Pressure Vessel Head Penetration Nozzles, Revision 1," dated September 2004 (ADAMS Accession No. ML043200602), and ongoing work with the Office of Nuclear Reactor Research. The results of the NRC staff review supports the licensee's stress analysis, and the NRC staff finds the areas of missed inspection coverage are in a reduced stress area, less than 20 ksi.

The licensee's fracture mechanics analysis showed that a conservative through wall axial flaw located in the uninspected region of the nozzle would not grow to the toe of the J-groove weld, in this case the edge of the reactor coolant pressure boundary, in less than 4 years of full-power operation. As the re-inspection frequency of each penetration nozzle at BVPS-2 is every refueling outage, the licensee's fracture mechanics analysis provides a basis for reasonable assurance of the structural integrity of each penetration nozzle with reduced inspection coverage as identified in Table 1 above. The NRC staff's assessment of the licensee's fracture mechanics analysis conclusions is based on data analysis of the supporting Figures B-1 through B-9 of the crack growth predictions for various nozzle angles, as provided in the licensee's submittal. In addition, the NRC staff performed an independent crack growth calculation, the results of which support the licensee's analysis. Therefore, the NRC staff concurs with the adequacy of the re-inspection frequency to provide reasonable assurance of structural integrity of each nozzle due to the area of missed inspection coverage as defined in Table 1 above.

The safety issues that are addressed by the 10 CFR 55a(g)(6)(ii)(D) are degradation of the low-alloy steel RPV upper head, reactor coolant pressure boundary integrity and ejection of the RPV upper head penetration nozzle due to circumferential cracking of the nozzle above the J-groove weld. The licensee's proposed alternative inspection provides reasonable assurance that these safety issues are addressed at BVPS-2. The licensee has noted that while surface examination could be performed to increase the inspection coverage for the nozzle, these additional inspections would be of limited value and require extensive work in very high radiation fields. The NRC staff finds that performing these additional surface examinations would result in hardship through significant radiation exposure without a compensating increase in the level of quality or safety.

## 4.0 CONCLUSION

Based on the above discussion, the NRC staff has concluded that compliance with 10 CFR 50.55a(g)(6)(ii)(D) would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, and that the proposed alternative provides reasonable assurance of structural integrity. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff authorizes the proposed alternative for BVPS-2 for the third 10-year ISI interval, or until the reactor vessel head is replaced, whichever occurs first.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principle Contributor: J. Collins

Date: September 28, 2009

If you have any questions, please contact the Beaver Valley Project Manager, Nadiyah Morgan, at (301) 415-1016.

Sincerely,

/RA/

Nancy L. Salgado, Chief
Plant Licensing Branch 1-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-412

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